Experimental Study of Integral Data of BFS Fast Critical Assembly with Lead Coolant

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So as the analysis of the experiments has shown that the use of neutron data from files of various national libraries results in significant differences in the calculated values of basic neutronic characteristics, a series of critical assemblies at the BFS facility was created for the validation of the neutron data for Pb and Bi in fast reactor spectrum [1].

The experimental program aimed to the measurements of the integral parameters such as spectral indices, capture in uranium vs. fission of plutonium, central coefficients of reactivity, void effect of coolant was carried out on BFS-77 critical assembly. This assembly was the benchmark type model of BREST-300 reactor (mixed nitride fuel, lead coolant, BR for the core of ≈ 1). The results of the experiments, so as the assembly description prepared for the analysis.

Calculations of the measured parameters are carried out using Russian calculation codes and the most modern versions of nuclear data libraries (ABBN-93, ENDF/B-6, JENDL-3.2 etc.).

Calculation benchmark models of the experiment are constructed. They are used for testing of precise (with detailed description of geometry and neutrons transition) and engineer (with homogeneous core description and diffusion theory of neutrons transition) neutron physics codes and integral neutron data. Correction factors necessary to adequate comparison with the measurements are obtained for a simpler model and their uncertainty are estimated. Corresponding recommendations on their uncertainties and/or correction will be worked out.

References

1. A. Kochetkov et. al. "Validation of Neutron Data for Pb and Bi Using Critical Experiments" - ND-2001, Tsukuba, Japan.

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